By adhering to the SRD development process, the SRD demonstrates, that when implemented, the set of standards contained in Volume II provides

1) Adequate safety in light of the hazards posed by facility operations

2) Compliance with applicable laws and regulations

3) Conformance to DOE top-level safety standards and objectives as specified in DOE/RL-96-0006 *Top-Level Radiological, Nuclear, and Process Safety Standards and Principles for TWRS Privatization Contractors* (DOE-RL 1996d)

4) Consideration of experience embodied in commercial codes and standards.

The following chapters describe the standards identification process used in development of the SRD. This process is broken down into process initiation, identification of work, hazards assessment, identification of standards, and confirmation of adequacy. Attachments A, B, and C document the qualification of the participants in the SRD development process. Attachment D is a listing and summary of the laws, regulations, and industry standards used in the development of the SRD. Attachment E is a matrix demonstrating compliance with applicable laws and contractual requirements. Attachment F of Revision 0 has been replaced with "Radiological Exposure Standards for the TWRS-P Project~~.~~" as Appendix D in Volume II of the SRD. The original Attachment F supported the initial review of the SRD by mapping the attributes of DOE/RL-97-08, *Guidance for the Review of TWRS Privatization Contractor Safety Requirements Document Submittal Package,* DOE-RL 1997, to the Standards Approval Package submittals incorporating the expectations as stated in the attributes. ~~Attachment F~~SRD Volume II, Appendix D, "Radiological Exposure Standards for the TWRS-P Project," incorporates the stand-alone document *Radiological and Nuclear Exposure Standards for Facility and Co-Located Workers* (BNFL 1997c). The attachment has been revised and updated to reflect the BNFL Inc. responses to DOE Regulatory Unit questions on the Standards Approval Package and Initial Safety Assessment.

1.0 Radiological, Nuclear and Process Safety Objectives

Safety Criterion: 1.0 - 1

A comprehensive radiological and process safety management program shall be used to eliminate or reduce the incidence, or mitigate the consequences of, accidental radioactive or chemical releases, process fires, and process explosions. This program shall address management practices, technologies, and procedures. Radiological and process safety management shall confirm that the facility is properly designed, the integrity of the design is maintained, and the facility is operated according to the safe manner intended.

Implementing Codes and Standards:

BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan
Section: 4.1 Safety Management Processes
Chapter: 5.0 Process Safety Management

Regulatory Basis:

29 CFR 1910 Occupational Safety and Health Standards Location: 119
40 CFR 68 Chemical Accident Prevention Provisions Location: 10
DOE/RL-96-0006 5.1.1 Process Safety Management
DOE/RL-96-0006 5.1.2 Process Safety Objective

Safety Criterion: 1.0 - 2

Principal emphasis shall be placed on the prevention of accidents, particularly any that could cause an unacceptable release, as the primary means of achieving safety.

Implementing Codes and Standards:

BNFL-5193-SRD-01, Appendix A Implementing Standard for ~~Defense-In-Depth~~Safety Standards and Requirements Identification
DOE IG Implementation Guide for Nonreactor Nuclear Safety Design Criteria and Explosive Safety Criteria, 2.3
DOE Order 420.1 Facility Safety, 4.1.1.2

Regulatory Basis:

DOE/RL-96-0006 4.1.1.2 Defense in Depth-Prevention

Safety Criterion: 1.0 - 3

The risk, to an average individual within 1 mile of the TWRS-P Controlled Area Boundary, of prompt fatalities that might result from an accident shall not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents which members of the U.S. population generally are exposed.

Implementing Codes and Standards:

~~BNFL-5193-SRD-01, Attachment F Radiological Exposure Standards For The TWRS-P Project~~
~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~
~~Section 1.3.8 Acceptable Level of Worker Safety~~
~~Chapter 12.0 Definitions "Controlled Area"~~
BNFL-5193-SRD-01, Appendix A, Implementing Standard for Safety Standards and Requirements Identification
BNFL-5193-SRD-01, Appendix D, Radiological Exposure Standards for the TWRS-P Project

Regulatory Basis:

DOE/RL-96-0006 3.1.2 Accident Risk Goal

Safety Criterion: 1.0 - 4

The risk, to the public and workers within 16 km (10 miles) of the TWRS-P Facility, of cancer fatalities that might result from TWRS-P facility operations shall not exceed one-tenth of one percent (0.1%) of the sum of cancer fatality risks to which members of the U.S. population generally are exposed.

Implementing Codes and Standards:
~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~
~~Section 3.6.1 Normal Operations~~
BNFL-5193-SRD-01, Appendix A, Implementing Standard for Safety Standards and Requirements Identification
BNFL-5193-SRD-01, Appendix D, Radiological Exposure Standards for the TWRS-P Project

Regulatory Basis:
DOE/RL-96-0006 3.1.1 Operations Risk Goal

Safety Criterion: 1.0 - 5

The risk to workers within the BNFL TWRS-P Controlled Area Boundary, of fatality from radiological exposure that might result from an accident, shall not be a significant contributor to the overall occupational risk of fatality to workers.

Implementing Codes and Standards:
~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~
~~Section: 1.3.8 Acceptable Level of Worker Safety~~
BNFL-5193-SRD-01, Appendix A, Implementing Standard for Safety Standards and Requirements Identification
BNFL-5193-SRD-01, Appendix D, Radiological Exposure Standards for the TWRS-P Project

Regulatory Basis:
DOE/RL-96-0006 3.1.3 Worker Accident Risk Goal

Safety Criterion: 1.0 - 6

Measures in the design and operation of the facility to protect the public, workers, and environment against accident conditions shall be evaluated using an acceptable approach to demonstrate that they perform their intended purpose with high confidence.

Implementing Codes and Standards:
BNFL-5193-SRD-01, Appendix B Implementing Standard for Defense in Depth
~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~
~~Section: 1.3.11 Quality Levels~~
~~Section: 3.6 Facility Design for Postulated Events~~
~~Section: 3.7 Proven Engineering Practices~~
~~Section: 3.11 Safety Systems Design~~
BNFL-5193-SRD-01, Appendix A, Implementing Standard for Safety Standards and Requirements Identification

Regulatory Basis:
DOE/RL-96-0006 3.3.1 Public Protection
DOE/RL-96-0006 3.3.2 Worker Protection

Safety Criterion: 1.0 - 7

To compensate for potential human and equipment failures, a defense-in-depth strategy shall be applied to the facility commensurate with the hazards; such that, as appropriate to control the risk, safety is vested in multiple, independent safety provisions, no one of which is to be relied upon excessively to protect the public, the workers, or the environment. This strategy shall be applied to the design and operation of the facility.

Implementing Codes and Standards:

ANSI/ANS 58.9-1981 Single Failure Criteria for Light Waster Reactor Safety-Related Fluid Systems
BNFL-5193-SRD-01, Appendix B Implementing Standard for Defense in Depth
DOE IG Implementation Guide for Nonreactor Nuclear Safety Design Criteria and Explosive Safety Criteria, 2.3
DOE Order 420.1 Facility Safety 4.1.1.2
IEEE 379-1994 Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems

Regulatory Basis:

DOE/RL-96-0006 4.1.1.1 Defense in Depth-Defense in Depth

Safety Criterion: 1.0 - 8

Structures, systems, and components (SSCs) that serve to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the workers and the public are classified as Important to Safety. It encompasses the broad class of facility features addressed (not necessarily explicitly) in the top-level radiological, nuclear, and process safety standards and principles that contribute to the safe operation and protection of workers and the public during all phases and aspects of facility operations (i.e., normal operation as well as accident mitigation). This definition includes not only those structures, systems, and components that perform safety functions and traditionally have been classified as safety class, safety-related or safety-grade, but also those that place frequent demands on or adversely affect the performance of safety functions if they fail or malfunction, i.e., support systems, subsystems, or components. Thus, these latter structures, systems, and components would be subject to applicable top-level radiological, nuclear, and process safety standards and principles to a degree commensurate with their contribution to risk. In applying this definition, it is recognized that during the early stages of the design effort all significant systems interactions may not be identified and only the traditional interpretation of Important to Safety, i.e., safety-related may be practical. However, as the design matures and results from risk assessments identify vulnerabilities resulting from non-safety-related equipment, additional structures, systems, and components should be considered for inclusion within this definition.

Important to Safety includes SSCs designated as Safety Design Class and Safety Design Significant.

Safety Design Class SSCs includes those that, by performing their specified safety function, prevent workers or the maximally exposed member of the public from receiving a radiological exposure that exceeds the exposure standards defined in the SRD. Safety Design Class also applies to those features that by functioning, prevent the worker or maximally exposed member of the public from receiving a chemical exposure that exceeds the ERPG-2 (AIHA 1988) chemical release standard. Those features credited for the prevention of a criticality event are also designated as Safety Design Class.

Safety Design Significant SSCs are those needed to achieve compliance with the radiological or chemical exposure standards for the public and workers during normal operation; and SSCs that can, if they fail or malfunction, place frequent demands on, or adversely affect the function of, Safety Design Class SSCs.

Implementing Codes and Standards:

~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~
~~Section: 1.3.7 Acceptable Level of Public Safety~~
~~Section: 1.3.8 Acceptable Level of Worker Safety~~
~~Section: 1.3.10 Classification of Structures, Systems, and Components~~
~~BNFL-5193-SRD-01, Appendix A, Implementing Standard for Safety Standards and Requirements Identification~~
~~BNFL-5193-SRD-01, Appendix D, Radiological Exposure Standards for the TWRS-P Project~~

Regulatory Basis:

DOE/RL-96-0006 3.3.1 Public Protection
DOE/RL-96-0006 3.3.2 Worker Protection

Safety Criterion: 1.0 - 9

BNFL Inc. shall accept ultimate responsibility for the safety of the TWRS-P Facility.

Implementing Codes and Standards:

BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan
Chapter: 1.0 Safety Approach to TWRS Privatization
Section: 11.1 Design and Construction Phase
Section: 11.2 Operations Phase

Regulatory Basis:

DOE/RL-96-0006 4.1.2.1 Safety Responsibility-Safety Responsibility
DOE/RL-96-0006 4.3.1.1 Conduct of Operations-Organizational Structure
DOE/RL-96-0006 5.1.3 Process Safety Responsibility

Safety Criterion: 1.0 - 10

In addition to the Safety Criteria contained herein, compliance with all requirements of 10 CFR 830.120 and 10 CFR 835 shall be achieved absent the granting of an exemption request to any specific requirement therein.

Implementing Codes and Standards:

~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~
~~Chapter: 2.0 Compliance with Laws and Regulations~~
~~Section: 2.2 Compliance with 10 CFR 830.120, "Quality Assurance Requirements"~~
~~Section: 2.3 Compliance with 10 CFR 835, "Occupational Radiation Protection"~~

Regulatory Basis:

10 CFR 830.120 Quality assurance requirements Location:
10 CFR 835 Occupational Radiation Protection Location: 1
DE-AC06-96RL13308 Part I Section C.5 Table S4-1

2.0 Radiological and Process Standards

Safety Criterion: 2.0 - 1

The following Radiological Dose Standards shall be applied to protect the public and workers from TWRS-P radiological hazards.

Table 2-1. Radiological Exposure Standards Above Normal Background.

Description	Estimated Frequency of Occurrence f (yr^{-1})	General Guidelines	Worker	Co-located Worker	Public
Normal Events: Events that occur regularly in the course of facility operation (e.g., normal facility operations); including routine and preventive maintenance activities.	>0.1	Normal modes of operating facility systems should provide adequate protection of health and safety.	≤5 rem/yr ≤50 rem/yr any organ, skin, or extremity ≤15 rem/yr lens of eye ≤1.0 rem/yr ALARA design objective per 10CFR835.1002(b)[1]	≤5 rem/yr ≤1.0 rem/yr ALARA design objective per 10 CFR 835.1002(b)[1]	≤10 mrem/yr (airborne pathway) ≤100 mrem/yr (all sources) ≤100 mrem/yr (public in the controlled area) ≤25 mrem/yr (radioactive waste)
Anticipated Events: Events of moderate frequency that may occur once or more during the life of a facility (e.g., minor incidents and upsets).	$10^{-2}<f\leq10^{-1}$	The facility should be capable of returning to operation without extensive corrective action or repair.	≤5 rem/event [2,3] 1.0 rem/event design action threshold [4]	≤5 rem/event (2, 3) 1.0 rem/event design action threshold[4]	≤100 mrem/event[3]
Unlikely Events: Events that are not expected, but may occur during the lifetime of a facility (e.g., more severe incidents).	$10^{-4}<f\leq10^{-2}$	The facility should be capable of returning to operation following potentially extensive corrective action or repair, as necessary.	≤25 rem/event[2,3]	≤25 rem/event[2, 3]	≤5 rem/event[3]
Extremely Unlikely Events: Events that are not expected to occur during the life of the facility but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material.	$10^{-6}<f\leq10^{-4}$	Facility damage may preclude returning to operation.	≤25 rem/event[2,3]	≤25 rem/event[2,3]	≤25 rem/event ≤5 rem/event target[3] ≤300 rem/event to thyroid
Location of Receptor			Within the BNFL TWRS-P Controlled Area Boundary, including 241-AP-106	The most limiting location at or beyond the BNFL TWRS-P Controlled Area Boundary	The most limiting location along the near river bank/Hwy240 /southern boundary

3.3 Criticality

Safety Criterion: 3.3 - 1

The facility shall be designed and operated in a manner that prevents nuclear criticality.

Implementing Codes and Standards:

ANSI/ANS 8.1-1983 (R 1988) Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors

ANSI/ANS 8.19-1996 Administrative Practices for Nuclear Criticality Safety

Regulatory Basis:

DOE/RL-96-0006 4.2.2.5 Proven Engineering Practices/Margins-Criticality

Safety Criterion: 3.3 - 2

The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations and in the nature of the immediate environment under accident conditions.

The multiplication factor (k<eff>), including all biases and uncertainties at a 95% confidence level, shall be shown to not exceed 0.95 under all credible normal, off-normal, and accident conditions.

OR

The evaluated parameter (i.e., effecting the margin of subcriticality) is equal to or less than a corresponding subcritical limit given in accepted ANSI/ANS-8 standards or guides that are selected for use in the RPP-WTP Project.

~~**Implementing Codes and Standards:**~~

~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~

~~Section: 3.8 Criticality Safety~~

Safety Criterion: 3.3 - 3

Process designs shall incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible. Protection shall be provided by either :

(1) the control of two independent process parameters (which is the preferred approach, when practical, to prevent common-mode failure), or

(2) a system of multiple controls on a single process parameter.

The number of controls required for a single controlled process parameter shall be based upon control reliability and any features that mitigate the consequences of control failure. In all cases, no single credible event or failure shall result in the potential for a criticality accident.

An exception to the application of double contingency, where single contingency operations are permissible, is presented in paragraph 5.1 of ANSI/ANS-8.10-1983, R88. This exception applies to operations with shielding and confinement (e.g., hot cells or other shielded facilities).

Double contingency shall be demonstrated by documented evaluations.

~~**Implementing Codes and Standards:**~~

~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~

~~Section: 3.8 Criticality Safety~~

3.3 Criticality

Safety Criterion: 3.3 - 1

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Regulatory Basis:

DOE/RL-96-0006 4.2.2.5 Proven Engineering Practices/Margins-Criticality

Safety Criterion: 3.3 - 2

The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations and in the nature of the immediate environment under accident conditions.

The multiplication factor (k<eff>), including all biases and uncertainties at a 95% confidence level, shall be shown to not exceed 0.95 under all credible normal, off-normal, and accident conditions.

OR

The evaluated parameter (i.e., effecting the margin of subcriticality) is equal to or less than a corresponding subcritical limit given in accepted ANSI/ANS-8 standards or guides that are selected for use in the RPP-WTP Project.

~~**Implementing Codes and Standards:**~~

~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~

~~Section: 3.8 Criticality Safety~~

Safety Criterion: 3.3 - 3

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The number of controls required for a single controlled process parameter shall be based upon control reliability and any features that mitigate the consequences of control failure. In all cases, no single credible event or failure shall result in the potential for a criticality accident.

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Double contingency shall be demonstrated by documented evaluations.

~~**Implementing Codes and Standards:**~~

~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~

~~Section: 3.8 Criticality Safety~~

Safety Criterion: 3.3 - 4

Where a sufficient quantity of fissionable material is being processed such that criticality safety is a concern, passive engineered controls, such as geometry control, shall be considered as the preferred control method. Where passive engineered control is not feasible, the preferred order of controls is active engineered controls followed by administrative controls. The double contingency analysis shall justify the chosen controls. Full advantage may be taken of any nuclear characteristics of the process materials and equipment. The geometry must be considered as water moderated and reflected unless it can be shown the presence of water is not credible. All dimensions, nuclear properties, and other features upon which reliance is placed shall be documented and verified prior to beginning operations, and control shall be exercised to maintain them.

~~Implementing Codes and Standards:~~
~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~
~~Section: 3.8 Criticality Safety~~

Safety Criterion: 3.3 - 5

To protect against an uncontrolled nuclear criticality incident, nuclear criticality safety considerations and controls shall be evaluated for accidents, normal operations, and before any significant operational changes are made.

~~Implementing Codes and Standards:~~
~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~
~~Section: 3.8 Criticality Safety~~

Safety Criterion: 3.3 - 6

Criticality Accident Alarm Systems (CAS) and Criticality Detection Systems (CDS) shall be required as follows:

(1) In those locations where the mass of fissionable material exceeds the limits established in Table 3-1 Inventory of Fissionable Material and the probability of a criticality accident is greater than 1E-06 per year, a CAS conforming to ANSI/ANS-8.3-1986 shall be provided to cover occupied areas in which the expected dose exceeds 12 rads (0.12 greys) in free air, where a CAS is defined to include a criticality accident detection device and a personnel evacuation alarm.

(2) In those locations where the mass of fissionable material exceeds the limits established in Table 3-1 Inventory of Fissionable Material and the probability of a criticality accident is greater than 1E-06 per year, but there are no occupied areas in which the expected dose exceeds 12 rads (0.12 greys) in free air, a CDS shall be provided, where a CDS is defined to be an appropriate criticality accident detection device but without an immediate evacuation alarm. The CDS response time should be sufficient to allow for appropriate process-related mitigation and recovery actions. Appropriate response guidance to minimize personnel exposure shall be provided.

(3) In those locations where the mass of fissionable material exceeds the limits established in Table 3-1 Inventory of Fissionable Material, but a criticality accident is determined to be impossible due to the physical form of the fissionable material, or the probability of occurrence is determined to be less than 1E-06 per year, neither a CAS nor a CDS is required. Neither a CAS nor a CDS is required for fissionable material during shipment when packaged in approved shipping containers, or when packaged in approved shipping containers awaiting transport provided that no other operation involving fissionable material not so packaged is permitted on the shipping dock or in the shipment area.

(4) If a criticality accident is possible wherein a slow (i.e., quasistatic) increase in reactivity could occur leading from subcriticality to criticality to self-shutdown without initiating emplaced criticality alarms, CASs should be supplemented by warning devices such as audible personnel dosimeters (e.g., pocket chirpers/flashers, or their equivalents), area radiation monitors, area dosimeters, or integrating CASs to aid in protecting workers against the consequences of slow criticality accidents.
(5) Neither a CAS nor a CDS is required to be installed for handling or storage of fissionable material when sufficient shielding exists that is adequate to protect personnel (e.g., hot cells); however, a means to detect fission product gases or other volatile fission products shall be provided in occupied areas immediately adjacent to such shielded areas, except for systems where no fission products are likely to be released.

Note: The frequency of 1E-06 per year is used as a measure of credibility and does not require a probabilistic risk assessment be performed. Reasonable grounds for incredibility may be presented on the basis of commonly accepted engineering judgement.

Table 3-1 Inventory of Fissionable Material[1]

Isotope	Inventory in Individual Unrelated Area
U-235	700g
U-233	520g
Pu-239	450g
Any Combination of above Isotopes	450g

[1] Per ANSI/ANS-8.3-1986 paragraph 4.2.1

~~Implementing Codes and Standards:~~
~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~
~~Section: 3.8 Criticality Safety~~

Safety Criterion: 3.3 - 7

The monitoring system shall be capable of detecting a criticality that produces an absorbed dose in soft tissue of 20 rads (0.20 greys) of combined neutron and gamma radiation at an unshielded distance of 2 meters from the reacting material within one minute.

~~Implementing Codes and Standards:~~
~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~
~~Section: 3.8 Criticality Safety~~

Safety Criterion: 3.3 - 8

Coverage of all areas requiring detection may be provided by a single detector.

~~Implementing Codes and Standards:~~
~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~
~~Section: 3.8 Criticality Safety~~

4.1 General Design

Safety Criterion: 4.1 - 1

The facility design shall provide for the prevention and mitigation of the risks associated with radiological and chemical material inventories and energy sources. The facility design shall include consideration of normal operation (including startup, testing and maintenance), anticipated operational occurrences, external events, and accident conditions.

Prevention shall be the preferred means of achieving safety.

Defense-in-depth shall be applied commensurate with the hazard to provide multiple physical and administrative barriers against undue radiation and chemical exposure to the public and workers.

Implementing Codes and Standards:

ANSI/ANS 58.9-1981 Single Failure Criteria for Light Waster Reactor Safety-Related Fluid Systems
BNFL IS-DID Implementing Standard for Defense in Depth
DOE IG Implementation Guide for Nonreactor Nuclear Safety Design Criteria and Explosive Safety Criteria, 2.3
DOE Order 420.1 Facility Safety 4.1.1.2
IEEE 379-1994 Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems

Regulatory Basis:

DOE/RL-96-0006 4.1.1.1 Defense in Depth-Defense in Depth
DOE/RL-96-0006 4.1.1.2 Defense in Depth-Prevention
DOE/RL-96-0006 4.2.1.1 Design-Safety Design

Safety Criterion: 4.1 - 2

Structures, systems, and components designated as Important to Safety shall be designed, fabricated, erected, constructed, tested, inspected, and maintained to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components designated as Important to Safety shall be maintained through deactivation of the facility.

Items and processes shall be designed using sound engineering/scientific principles and appropriate standards.

Design features that enhance the margin of safety through simplified, inherently safe, passive, or other highly reliable means to accomplish the specified safety function should be employed to the maximum extent practical.

Design work, including changes, shall incorporate applicable requirements and design bases. Design interfaces shall be identified and controlled. The adequacy of design products shall be verified or validated by individuals or groups other than those who performed the work. Verification and validation work shall be completed before approval and implementation of the design.

Safety technologies incorporated into the facility design should have been proven by experience or testing and should be reflected in approved codes and standards. Significant new design features should be introduced only after thorough research and model or prototype testing at the component, system, or facility level, as appropriate, to achieve the necessary level of confidence that the design feature will perform as expected.

Implementing Codes and Standards:

ACI 318-95 Building Code Requirements for Structural Concrete
ACI 318R-95 Commentary on Building Code Requirements for Structural Concrete
ACI 349-90 Code Requirements for Nuclear Safety-Related Concrete Structures
ACI 349R-90 Commentary on Code Requirements for Nuclear Safety-Related Concrete Structures
AISC MO16-89 Manual for Steel Construction - Allowable Stress Design, Ninth Edition
AISC N690-84 Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities
ASCE 4-86 Seismic Analysis of Safety-Related Nuclear Structures and Commentary
ASCE 7-95 Minimum Design Loads for Buildings and Other Structures
DOE-STD 1020-94 (Change 1, 1996) Natural Hazards Design and Evaluation Criteria for Department of Energy Facilities
DOE-STD-1021-93 (Change 1, 1996) Natural Phenomena Hazards Performance Categorization Guidelines for Structures, Systems, and Components
UBC Uniform Building Code

BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan
~~Section: 1.3.10 Classification of Structures, Systems, and Components~~
~~Section: 1.3.11 Quality Levels~~
Section: 1.3.16 Configuration Management
~~Section: 3.7 Proven Engineering Practices~~
~~Section: 3.11 Safety Systems Design~~
Section: 5.3 Configuration Management
<u>BNFL-5193-SRD-01, Appendix A, Implementing Standard for Safety Standards and Requirements Identification.</u>

<u>Regulatory Basis:</u>

DOE/RL-96-0006 4.1.2.4 Safety Responsibility-Operating Experience and Safety Research
DOE/RL-96-0006 4.1.5.1 Configuration Management-Formal Configuration Management
DOE/RL-96-0006 4.1.6.2 Quality Assurance-Established Techniques and Procedures
DOE/RL-96-0006 4.2.2.1 Proven Engineering Practices/Margins-Proven Engineering Practices
DOE/RL-96-0006 4.2.2.3 Proven Engineering Practices/Margins-Safety System Design and Qualification
DOE/RL-96-0006 4.2.5.1 Inherent/Passive Safety Characteristics-Safety Margin Enhancement

Safety Criterion: 4.1 - 3

This criterion addresses natural phenomena hazards (NPH) design for structures, systems, and components (SSCs) that are Important to Safety and have NPH safety functions.

SSCs designated as Important to Safety (i.e., Safety Design Class and Safety Design Significant) shall be designed to withstand the effects of NPH events such as earthquakes, wind, and floods without loss of capability to perform specified safety functions required as the result of the NPH events. This includes both the front line and support systems that must function for a NPH event such that the public or worker exposure standards of Safety Criterion 2.0-1 or 2.0-2 are not exceeded.

SSCs that are designated Safety Design Class and that are required to perform a safety function as a result of a given NPH shall be designed to withstand the NPH loadings of that NPH as provided in Table 4-1.

SSCs that are designated Safety Design Significant whose continued function is not required for an NPH event, but whose failure as a result of an NPH event could reduce the functioning of a Safety Design Class SSC such that exposure standards might be exceeded, shall be designed to withstand the NPH loadings of that NPH as provided in Table 4-1. For these SSCs, however, for seismic response only, credit may be taken for inelastic energy absorption per Table 2-4 of DOE-STD-1020-94.

For any SSC included under this criterion, other NPH loads (for which the SSC has no safety function) may be taken from Safety Criterion 4.1-4 and Table 4-2 in lieu of Safety Criterion 4.1-3 and Table 4-1.

Table 4-1 Natural Phenomena Design Loads for Important to Safety SSCs with NPH Safety Functions

Hazard	Load	Application documents
Seismic	Equal-hazards response spectra[a] 0.24 *g* horizontal, @ 33 hz 0.16 *g* vertical, @ 50 hz See Figure 4-1	DOE-STD-1020-94[b]
Straight wind	49.6 m/s (111 mi/hr), 3-second gust, at 10 m (33 ft) above ground, Importance factor, I=1.0	ASCE-7-95[c] DOE-STD-1020-94[b]
Wind Missile	5 cm x 10 cm (2x4) timber plank, 6.8 Kg (15 lb) at 22 m/s (50 mi/hr) (horiz), Max height 9 m. (30 ft)	DOE-STD-1020-94[b]
Tornado and Tornado Missiles	Not Applicable	DOE-STD-1020-94[b]
Volcanic ash	61 kg/m^2 (12.5 lbm/ft^2) ground ash load[d]	DOE-STD-1020-94[b]
Flooding	Dry site for river flooding Site drainage: 10 cm (3.9 in) for 6-hr precipitation	DOE-STD-1020-94[b]
Snow	75 kg/m^2 (15.0 lbm/ft^2) ground load	ASCE-7-95[c]

[a]Geomatrix, 1996, *Probabilistic Seismic Hazard Analysis DOE Hanford Site, Washington*, WHC-SD-W236A-TI-002, Rev.1, prepared for Westinghouse Hanford Company, Richland, Washington.
[b]DOE STD-1020-94,(1996)) *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*, U.S. Department of Energy, Washington, D.C.
[c]ASCE, 1995, *Minimum Design Loads for Building and Other Structures*, ASCE-7-95, American Society of Civil Engineers, New York, New York.
[d]WHC-SD-GN-ER-501, Rev.0, "Natural Phenomena Hazards, Hanford Site, South-Central Washington," Westinghouse Hanford Company.

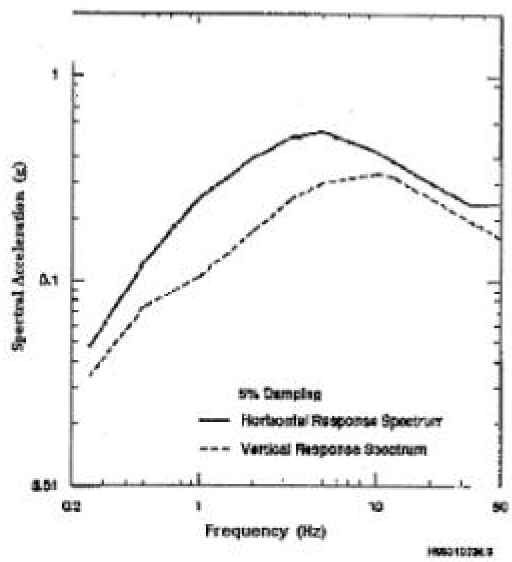

Figure 4-1 Spectral Acceleration for Important to Safety SSCs with NPH Safety Functions

Implementing Codes and Standards:

ACI 349-90 Code Requirements for Nuclear Safety-Related Concrete Structures
ACI 349R-90 Commentary on Code Requirements for Nuclear Safety-Related Concrete Structures
AISC N690-84 Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities
ASCE 4-86 Seismic Analysis of Safety-Related Nuclear Structures and Commentary
ASCE 7-95 Minimum Design Loads for Buildings and Other Structures
DOE-STD 1020-94 (Change 1, 1996) Natural Hazards Design and Evaluation Criteria for Department of Energy Facilities
DOE-STD-1021-93 (Change 1, 1996) Natural Phenomena Hazards Performance Categorization Guidelines for Structures, Systems, and Components
IEEE 344-1987 Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
UBC Uniform Building Code

~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~
~~Section: 1.3.10 Classification of Structures, Systems, and Components~~
BNFL-5193-SRD-01, Appendix A, Implementing Standard for Safety Standards and Requirements Identification.

Regulatory Basis:

DOE/RL-96-0006 4.2.2.2 Proven Engineering Practices/Margins-Common-Mode/Common-Cause Failure

Safety Criterion: 4.1 - 4

This criterion addresses natural phenomena hazards (NPH) design for structures, systems, and components (SSCs) without NPH safety functions.

SSCs that may be important to the safety of the TWRS-P Facility shall be designed to withstand the effects of NPH such as earthquakes, wind, and floods. The SSCs included under this criterion are:
1. SSCs Important to Safety (either Safety Design Class or Safety Design Significant) that do not have an NPH safety function; and
2. SSCs that are not Important to Safety and that have significant inventories of radioactive or hazardous materials but in amounts less than quantities that might lead to an Important to Safety designation.

SSCs included under this criterion shall be designed to withstand the NPH loadings as provided in Table 4-2.

Table 4-2 Natural Phenomena Design Loads for SSCs without NPH Safety Functions

Hazard	Load	Application documents
Seismic	Uniform Building Code, except as follows: ZC = 0.55 Importance Factor, I=1.25 R_w per Table 2-2 of DOE-STD-1020-94[b]	DOE-STD-1020-94[b]
Straight wind	38 m/s (85 mi/hr) 3-second gust, at 10 m (33 ft) above ground, Importance factor, I=1.07	ASCE-7-95[c] DOE-STD-1020-94[b]
Wind Missile	Not Applicable	DOE-STD-1020-94[b]
Tornado and Tornado Missiles	Not Applicable	DOE-STD-1020-94[b]
Volcanic ash	24 kg/m^2 (5 lbm/ft^2) ground ash load[d]	DOE-STD-1020-94[b]
Flooding	Dry site for river flooding Site drainage: 6.4 cm (2.5 in) for 6-hr precipitation[d]	DOE-STD-1020-94[b]
Snow	75 kg/m^2 (15.0 lbm/ft^2) ground load	ASCE-7-95[c]

[a]ICBO, 1994, *Uniform Building Code*, International Conference of Building Officials, Whittier, California.
[b]DOE STD-1020-94, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*, U.S. Department of Energy, Washington, D.C.
[c]ASCE, 1995, *Minimum Design Loads for Building and Other Structures*, ASCE-7-95, American Society of Civil Engineers, New York, New York.
[d]WHC-SD-GN-501, Rev.0, "Natural Phenomena Hazards, Hanford Site, South-Central Washington," Westinghouse Hanford Company

Implementing Codes and Standards:

ACI 318-95 Building Code Requirements for Structural Concrete
ACI 318R-95 Commentary on Building Code Requirements for Structural Concrete
AISC MO16-89 Manual for Steel Construction - Allowable Stress Design, Ninth Edition
ASCE 7-95 Minimum Design Loads for Buildings and Other Structures
DOE-STD 1020-94 (Change 1, 1996) Natural Hazards Design and Evaluation Criteria for Department of Energy Facilities
DOE-STD-1021-93 (Change 1, 1996) Natural Phenomena Hazards Performance Categorization Guidelines for Structures, Systems, and Components
UBC Uniform Building Code
~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~
~~Section: 1.3.10 Classification of Structures, Systems, and Components~~
BNFL-5193-SRD-01, Appendix A, Implementing Standard for Safety Standards and Requirements Identification.

Regulatory Basis:

DOE/RL-96-0006 4.2.2.2 Proven Engineering Practices/Margins-Common-Mode/Common-Cause Failure

4.2 Confinement Design

Safety Criterion: 4.2 - 1

The facility shall be designed to retain the radioactive and hazardous material through a conservatively designed confinement system for normal operations, anticipated operational occurrences, and accident conditions. The confinement system shall protect the worker and public from undue risk of releases such that the radiological and chemical exposure standards of Safety Criteria 2.0-1 and/or 2.0-2 are not exceeded.

Implementing Codes and Standards:

BNFL-5193-SRD-01, Appendix B Implementing Standard for Defense in Depth
BNFL-5193-SRD-01, Appendix A Implementing Standard for Safety Standards and Requirements Identification
DOE IG Implementation Guide for Nonreactor Nuclear Safety Design Criteria and Explosive Safety Criteria, 2.3
DOE Order 420.1 Facility Safety, 4.1.1.2

Regulatory Basis:

DOE/RL-96-0006 4.1.1.4 Defense in Depth-Mitigation

Safety Criterion: 4.2 - 2

Important to Safety liquid and gaseous systems and components, including pressure vessels, tanks, heat exchangers, piping, and valves, shall be designed to retain their hazardous inventory such that the radiological and chemical worker or public exposure standards of Safety Criteria 2.0-1 and/or 2.0-2 are not exceeded.

Implementing Codes and Standards:

ASME B31.3-96 Process Piping
ASME SEC VIII Boiler and Pressure Vessel Codes, Rules for Construction of Pressure Vessels
BNFL-5193-SRD-01, Appendix A Implementing Standard for Safety Standards and Requirements Identification
~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~
~~Section: 1.3.10 Classification of Structures, Systems, and Components~~

Safety Criterion: 4.2 - 3

Codes and standards for Important to Safety vessels and piping should be supplemented by additional measures (such as erosion/corrosion programs and piping in-service inspections) to mitigate conditions arising that could lead to a release of radiological or chemical material that would exceed the worker or public exposure standards of Safety Criteria 2.0-1 and/or 2.0-2.

Implementing Codes and Standards:

BNFL-5193-SRD-01, Appendix A Implementing Standard for Safety Standards and Requirements Identification
Document P001/2 Rules for the Design of Piping Systems
Document V001/2 Rules for the Design of Vessels

BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan
Section: 1.3.10 Classification of Structures, Systems, and Components
Section: 3.13 Reliability, Availability, Maintainability, and Inspectability (RAMI)
Section: 3.7.1 Passive Features

Regulatory Basis:

DOE/RL-96-0006 4.2.2.4 Proven Engineering Practices/Margins-Codes and Standards

Safety Criterion: 4.3 - 6

The possibility of human error in facility operations shall be taken into account in the design by facilitating correct decisions by operators and inhibiting wrong decisions and by providing means for detecting and correcting or compensating for error. The parameters to be monitored in control areas shall be selected and their displays arranged to ensure operators have clear and unambiguous indication of the status of the facility. The parameters and displays shall facilitate monitoring and the initiation and operation of systems designated as Important to Safety.

Implementing Codes and Standards:

BNFL-5193-SRD-01, Appendix B Implementing Standard for Defense in Depth
IEEE 1023-88 Guide for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations

Regulatory Basis:

DOE/RL-96-0006 4.1.1.6 Defense in Depth-Human Aspects
DOE/RL-96-0006 4.2.6.1 Human Factors-Human Error
DOE/RL-96-0006 4.2.6.2 Human Factors-Instrumentation and Control Design
DOE/RL-96-0006 4.2.6.3 Human Factors-Safety Status

Safety Criterion: 4.3 - 7

The control room or control area shall be designed to permit occupancy and actions to be taken to monitor the facility safely during normal operations, and to provide safe control of the facility for anticipated operational occurrences and accident conditions. If credit is taken for operator action to satisfy the accident exposure standards of Safety Criteria 2.0-1 and/or 2.0-2, adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body gamma and 30 rem beta skin for the duration of the accident. For occurrences and accidents involving chemical release, provisions shall be made such that the operator exposure does not exceed the worker exposure standards of Safety Criterion 2.0-2.

Consideration shall also be given to accidents at nearby facilities if operator action is required to safely control the processes and bring them to a safe state.

The need for an alternate system that would allow the processes to be placed in a safe state in the event the primary control area is uninhabitable shall be evaluated.

Implementing Codes and Standards:

ASME N509-89 Nuclear Power Plant Air Cleaning Units and Components
ASME N510-1989 (Rev 1995) Testing of Nuclear Air Cleaning Systems
NUREG-0800 Standard Review Plan, Section 6.4, Section II, Items 1-5.
~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan Section: 1.3.7 Acceptable Level of Public Safety~~
BNFL-5193-SRD-01, Appendix A, Implementing Standard for Safety Standards and Requirements Identification

Regulatory Basis:

DOE/RL-96-0006 4.2.4.1 Emergency Preparedness-Support Facilities
DOE/RL-96-0006 4.2.6.2 Human Factors-Instrumentation and Control Design

4.4 Electrical and Mechanical Systems

Safety Criterion: 4.4 - 1

A list of electric and mechanical components designated as Important to Safety shall be prepared and maintained. The list shall include:
(1) The performance specifications for normal operation and under conditions existing during and following accidents;
(2) The load, pressure, voltage, frequency, and other characteristics, as appropriate, for which the performance specified can be ensured.

Implementing Codes and Standards:
~~BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan~~
~~Section: 5.3 Configuration Management~~
BNFL-5193-SRD-01, Appendix A, Implementing Standard for Safety Standards and Requirements Identification

Safety Criterion: 4.4 - 2

Structures, systems, and components Important to Safety shall be designed and qualified to function as intended in the environments associated with the events for which they are intended to respond. The effects of aging on normal and abnormal functioning shall be considered in design and qualification.

Implementing Codes and Standards:
10 CFR 50.49 Environmental qualification of electric equipment important to safety for nuclear power
IEEE 323-83 Qualifying Class 1E Equipment for Nuclear Power Generating Stations

Regulatory Basis:

DOE/RL-96-0006 4.2.2.3 Proven Engineering Practices/Margins-Safety System Design and Qualification

Safety Criterion: 4.4 - 3

This Criterion has been deleted.

Safety Criterion: 4.4 - 4

Structures, systems, and components Important to Safety shall be designated, designed and constructed to permit appropriate inspection, testing, and maintenance throughout their operating lives to verify their continued acceptability for service with an adequate safety margin.

Systems and components designated as Important to Safety that are located in closed cells where access is not possible during facility operation or scheduled shutdown periods shall be designed and constructed to standards aimed at ensuring their suitability for the entire service life with an adequate safety margin. Alternately, provisions may be made for remote replacement, standby cells, or equipment or other methods capable of ensuring a serviceable facility with adequate safety for the duration of the intended operating life.

Implementing Codes and Standards:
BNFL-5193-SRD-01, Appendix A Implementing Standard for Safety Standards and Requirements Identification
IEEE 338-1987 Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems
IEEE 603-1991 Criteria for Safety Systems for Nuclear Power Generating Stations
BNFL-5193-ISP-01 TWRS-P Project Integrated Safety Management Plan
Section: 3.13 Reliability, Availability, Maintainability, and Inspectability (RAMI)

Regulatory Basis:

DOE/RL-96-0006 4.2.7.1 Reliability, Availability, Maintainability, and Inspectability(RAMI)-Reliability
DOE/RL-96-0006 4.2.7.2 Reliability, Availability, Maintainability, and Inspectability (RAMI)-Availability, Maintainability, and Inspectability

CONTENTS

4.3 ESTIMATION OF CONSEQUENCES

4.3.1 Accident Severity Level Identification

A severity level, SL, shall be assigned to each postulated radiological accident. The severity level shall reflect the unmitigated consequences of the postulated accident. Unmitigated consequences shall account for the quantity, form and location of the radioactive material available for release, and the energy sources available to interact with the hazardous material. Unmitigated consequences shall not account SSCs that serve to prevent or mitigate the release. Specifically, unmitigated consequences shall be evaluated on the basis of a ground level release. The severity level shall be defined as follows:

SL	Facility Worker Consequence	Co-Located Worker Consequence	Public Consequence
SL-1	> 25 rem/event	> 25 rem/event	> 5 rem/event
SL-2	5 - 25 rem/event	5 - 25 rem/event	1 - 5 rem/event
SL-3	1 - 5 rem/event	1 - 5 rem/event	0.1 – 1 rem/event
SL-4	< 1 rem/event	< 1 rem/event	< 0.1 rem/event

These severity levels are related to the radiological and process standards of SRD Section 2.0 as follows:

- The unmitigated consequences associated with SL-1 events exceed the radiological standards for extremely unlikely events (SRD Safety Criterion 2.0-1).
- The unmitigated consequences associated with SL-2 events are below the radiological standards for extremely unlikely events (SRD Safety Criterion 2.0-1).
- The unmitigated consequences associated with SL-3 events are below the radiological standards for unlikely events (SRD Safety Criterion 2.0-1).
- The unmitigated consequences associated with SL-4 events are below the radiological standards for anticipated events (SRD Safety Criterion 2.0-1).

Consequences to the facility worker shall be evaluated at the worst-case occupied location. Consequences to the co-located worker and the public shall be evaluated at the locations specified in ~~Attachment F~~Appendix D to the *Safety Requirements Document, Volume II*.

Early in the design, the severity level is estimated based on the experience of the hazard evaluation team. As the design progresses, these estimates are confirmed through the formal accident analyses described in Section 4.3.2. These accident analyses do not address all of the potential accidents identified, but they do address bounding examples of each type of accident. The team should use the results of the accident analyses to validate the severity level estimates for potential accidents not addressed in the formal accident analyses.

- Administrative controls (for example, limits on inventory).

Consistent with the defense in depth principle, the control strategy development should emphasize preventive measures. It should also emphasize passive SSCs over active SSCs and retention of released material over dispersion. Ideally, the preferred control strategy should incorporate SSCs that prevent releases and SSCs that mitigate the consequences of a release, should it occur.

Once the preferred control strategy is identified, it shall be evaluated using the techniques described in Section 4.3 through 4.5. In addition, the evaluation of the control strategy shall identify the measures necessary to assure that it performs its functions reliably. Such measures include maintenance requirements, testing intervals and calibration frequency. The results of this evaluation serve to confirm that the control strategy is capable of satisfying SRD Safety Criteria 2.0-1.

If credit is taken for operator action to satisfy the public radiological exposure standards of Safety Criterion 2.0-1, adequate radiation protection is provided to permit access and occupancy of the control room or other control locations under accident conditions without personnel receiving radiation doses in excess of 5 rem TEDE whole body gamma and 30 rem beta skin for the duration of the accident. If credit is taken for operator action to satisfy public chemical exposure to EPRG-2 limits, provisions for operational access and control are made so that the operator exposure does not exceed the EPRG-2 limits.

Documentation of the hazard control strategy development process shall be a narrative defining the overall approach to control a specific pre-identified hazard. The control strategy should be described in terms of the safety functions required (e.g., limit release of radionuclides, etc.) and in terms of a set of engineered features, administrative controls (procedures and training), and management systems selected for implementing the strategy. The documentation should identify all control strategies considered and provide a defensible rationale for selection of the preferred strategy.

The following information produced by the control strategy definition shall be recorded in the hazard database:

- Preferred control strategy
- Rationale for preferred control strategy selection
- Defense in depth provided
- Control strategy functions and performance requirements
- Estimate of the unmitigated event frequency
- Estimate of the consequences from the mitigated event
- Estimate of the mitigated event frequency
- Applicable design basis events (e.g., design basis earthquake)

This information in the hazard database links the specific hazards to specific control strategies.

One of the issues in developing a control strategy for a particular hazard is determining the number of layers of prevention and mitigation appropriate for the hazard. The control strategies shall conform to the requirements defined in the Implementing Standard for Defense in Depth. In addition, the following guidance shall be considered in developing control strategies.

The general TWRS-P design approach is to provide two confinement barriers against the release of hazardous materials. The process vessels and piping form the primary confinement barrier;

the process cells and associated ventilation system form the secondary confinement barrier. Releases from the primary confinement are mitigated by the secondary confinement.

The accident severity levels defined in Section 4.3.1 are related to the exposure standards in SRD Safety Criterion 2.0-1. The SRD Safety Criterion 2.0-1 exposure standards are frequency based, so it is possible to establish target frequencies for events with a given severity level. The target frequencies tabulated below are consistent with SRD Safety Criterion 2.0-1.

SL	Event Target Frequency (yr^{-1})
SL-1	$<10^{-6}$
SL-2	$<10^{-4}$
SL-3	$<10^{-2}$
SL-4	$<10^{-1}$

These target frequencies may be used to guide control strategy development as described below. For SL-1 events:

- Meeting the target frequency will usually require a control strategy that incorporates diverse and independent SSCs that act to prevent and mitigate the event.

- Meeting the target frequency ~~may~~ will usually require diverse SSCs that act to prevent the release.

- The degree of mitigation required depends on the release frequency, that is, on the reliability of the preventive SSCs. For example, assume that the preventive SSCs assure that the frequency of release is less than 10^{-4} per year, but more than 10^{-6} per year. This frequency is not acceptable for events that have SL-1 level consequences, but is acceptable for events that have SL-2 level consequences. Therefore, the control strategy would need to provide enough mitigation to reduce the consequences of the release to the levels associated with a SL-2 event, as a minimum. The combined reliability of the preventive SSCs and the SSCs that provide mitigation needs to satisfy the target frequency for a SL-1 event. That is, the probability that the SSCs that provide mitigation will fail should be on the order of 10^{-2}, given the release.

- SSCs in control strategies for SL-1 events shall satisfy the single failure criteria in the Implementing Standard for Defense in Depth.

For SL-2 events:

- Meeting the target frequency will usually require a control strategy that incorporates diverse and independent SSCs that act to prevent and mitigate the event.

- The degree of mitigation required depends on the release frequency, that is, on the reliability of the preventive SSCs. For example, assume that the only viable preventive SSCs assure that the frequency of release is less than 10^{-2} per year, but more than 10^{-4} per year. This frequency is not acceptable for events that have SL-2 level consequences, but is acceptable for events that have SL-3 level consequences. Therefore, the control strategy would need to provide enough mitigation to reduce the consequences of the release to the levels associated with a SL-3 event, as a minimum. The combined reliability of the preventive SSCs and the SSCs that provide mitigation needs to satisfy the target frequency for a SL-2 event. That is, the probability that the SSCs that provide mitigation will fail should be on the order of 10^{-2}, given the release.

- SSCs in control strategies for SL-2 events should satisfy the single failure criteria in the Implementing Standard for Defense in Depth.

For SL-3 and SL-4 events:

- The mitigation provided by the secondary confinement would be adequate to satisfy SRD Safety Criterion 2.0-1. It would also be adequate to satisfy SRD Safety Criteria 1.0-3 through 1.0-5. However, preventive features should be considered consistent with the defense in depth principle.

- A single preventive SSC may satisfy the frequency goal for SL-3 and SL-4 events.

- SSCs in control strategies for SL-3 and SL-4 events need not satisfy the single failure criteria in the Implementing Standard for Defense in Depth.

6.0 CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

The design classification process used on the TWRS-P Project provides a consistent, project-wide approach for the classification of the TWRS-P Facility SSCs based on their importance to controlling normal releases and accident prevention and mitigation. This approach ensures that SSCs are designed, constructed, fabricated, installed, tested, operated, and maintained to quality standards commensurate with the importance of the functions that need to be performed. As the facility moves to deactivation, and the safety functions change, the classification of SSCs can be revised as necessary.

BNFL Inc. has established a design classification system to provide assurance to DOE that the defined safety functions of SSCs will perform as intended.

SSCs defined as Important-to-Safety for the TWRS-P Facility include the following.

1) SSCs needed to prevent or mitigate accidents that could exceed public or worker radiological and chemical exposure standards of Safety Criteria 2.0-1 and 2.0-2 and SSCs needed to prevent criticality. This set of SSCs includes both the front line and support systems needed to meet these exposure standards or to prevent criticality. This set of Important-to-Safety SSCs are designated as Safety Design Class.

2) SSCs needed to achieve compliance with the radiological or chemical exposure standards for the public and workers during normal operation; and SSCs that place frequent demands on, or adversely affect the function of, Safety Design Class SSCs if they fail or malfunction. This set of Important-to-Safety SSCs are designated as Safety Design Significant.

   The processes for identifying the SSCs for each of the two groups of SSCs Important-to-Safety and the requirements assigned to each of the two groups are discussed below.

   Safety Design Class SSCs typically are identified by the results of accident analyses that show the potential for exposure standards to be exceeded or prevent a criticality. However, additional items may also be designated Safety Design Class independent of a specific accident analysis. These are items that protect the facility worker from potentially serious events. Typically, these events are deemed to present a challenge to the facility worker severe enough that mitigation is prudent, without the need to perform a specific consequence analysis.

   Safety Design Significant SSCs are identified in several ways including: (1) SSCs identified as significant contributors to safety by the analyses that confirm the facility accident risk goals are met (this is one way to identify SSCs that place frequent demands on, or adversely affect the function of, Safety Design Class SSCs if they fail or malfunction), (2) SSCs that are needed to ensure that standards for normal operation are not exceeded (e.g., bulk shield walls or radiation monitors), (3) SSCs selected based on the dictates of nuclear and chemical facility experience and prudent engineering practices, and (4) SSCs whose failure could prevent Safety Design Class SSCs from performing their safety function (e.g., Seismic II/I items).

When an SSC is designated as Safety Design Class it has the following attributes:

1) Quality Level 1 (QL-1) is applied to the SSC to provide added assurance that the SSCs can perform their specified safety function.

2) For an active system or component, the safety function is preserved by application of defense-in-depth such that failure of the system or component will not result in exceeding a public or worker accident exposure standard. For a mitigating feature, this means that, given that the accident has occurred, the consequence of the accident will not result in exceeding a public or worker exposure standard. For a preventative feature, this means that the failure of

the system or component will not allow the accident to occur and progress such that a public or worker accident exposure standard is exceeded. If the hazard analysis shows that these requirements are necessary, this requirement may be achieved by designing the Safety Design Class system or component to withstand a single active failure or by designating two separate and independent systems or components as Safety Design Class.

3) The SSC is designed to withstand the effects of natural phenomena such that it can perform any safety functions required as a result of a natural phenomena event in accordance with Safety Criterion 4.1-3.

4) General design requirements are applied as identified in Section 4.0 of the SRD for Safety Design Class SSCs.

5) Specific design requirements based on the type of component are applied as invoked in SRD Chapter 4.0.

6) Other design requirements may be applied based on the specific safety function to be performed by the Safety Design Class SSC. This specific safety function is determined from the accident analysis that identified the need for prevention or mitigation by Safety Design Class SSCs.

7) Operational requirements (e.g., periodic testing and preventative maintenance) are applied to Safety Design Class SSCs through the application of Technical Safety Requirements.

When an SSC is classified as Safety Design Significant it is has the following attributes.

1) Quality Level 2 (QL-2) is applied to the SSC to provide added assurance that the SSCs can perform their specified safety function.

2) The SSC is designed to withstand the effects of natural phenomena such that it can perform its safety functions required as a result of a natural phenomena event in accordance with Safety Criterion 4.1-4.

3) General and specific design requirements are applied as identified in Section 4.0 of the SRD for Safety Design Significant SSCs.

4) Other design requirements again may be applied based on the specific safety function to be performed by the Safety Design Significant SSC.

76.0 IDENTIFICATION OF STANDARDS

Identification of standards is an iterative activity. Initially, the set of standards and requirements is derived from a general understanding of the hazards inherent in the work. As the design evolves, the hazard evaluation and the development of the control strategies justify tailoring the set of standards to better fit the hazards.

The standards identification activity is performed by a team including work activity experts, hazard assessment experts, hazard control experts, and ESH standards experts. The aim of this activity is to identify a tailored set of standards and requirements that will assure adequate safety when implemented.

Work activity experts shall be drawn from the following TWRS-P organizations:

- Functional staff of the TWRS-P Engineering Manager
- Technical staff of the HLW and LAW Vitrification Project Design Manager
- Technical staff of the BOF and Pretreatment Project Design Manager

~~7~~8.0 CONFIRMATION OF STANDARDS

Based on the recommendation of the process manager, the TWRS-P Project Manager requests the Project Safety Committee (PSC) to confirm the selected set of standards. The PSC defines a review approach, carries out the review, and documents the findings of the review. Comments by the PSC shall receive formal disposition by the Process Management Team.

~~8~~9.0 FORMAL DOCUMENTATION

Following confirmation by the PSC, the standards selection process shall be described in the Integrated Safety Management Plan. The results of the process shall be documented in the Safety Requirements Document (SRD). The SRD shall incorporate documentation supporting these results by reference. The SRD shall identify and justify the set of requirements and standards selected to provide adequate protection of workers, the public, and the environment.

~~9~~10.0 RECOMMENDATION

The TWRS-P Manager of Operations certifies that the recommended set of standards, when properly implemented:

1) Provides adequate safety
2) Complies with applicable laws and regulations
3) Conforms with the Top-Level Safety Standards and Principles.

~~10~~11.0 DEFINITIONS

Credible event: Any event with a frequency greater than 10^{-6} per year, including allowance for uncertainties.

Important to Safety: Structures, systems and components that serve to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the workers and the public. It encompasses the broad class of facility features addressed (not necessarily explicitly) in the top-level radiological, nuclear, and process safety standards and principles that contribute to the safe operation and protection of workers and the public during all phases and aspects of facility operations (i.e., normal operation as well as accident mitigation).

This definition includes not only those structures, systems, and components that perform safety functions and traditionally have been classified as safety class, safety-related, or safety-grade, but also those that place frequent demands on or adversely affect the performance of safety functions if they fail or malfunction, i.e., support systems, subsystems, or components. Thus, these latter structures, systems, and components would be subject to applicable top-level radiological, nuclear, and process safety standards and principles to a degree commensurate with their contribution to risk. In applying this definition, it is recognized that during the early stages of the design effort all significant systems interactions may not be identified and only the traditional interpretation of important to safety, i.e., safety-related, may be practical. However, as the design matures and results from risk assessments identify vulnerabilities resulting from non-safety-related equipment, additional structures, systems, and components should be considered for inclusion within this definition.

Mitigated event: As used in this standard, a mitigated event involves the following sequence:

- An initiating event that could lead to a release from the primary confinement barrier
- Failure of all elements of the control strategy that would prevent the initiating event from developing into a release from the primary confinement barrier
- Mitigation of the consequences of the release as provided by the control strategy

Mitigated event frequency: The mitigated event frequency is the corresponding release frequency times the probability that the elements of the control strategy that mitigate the release will function given the release.

APPENDIX D

RADIOLOGICAL EXPOSURE STANDARDS FOR THE TWRS-P PROJECT

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CONTENTS

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1.0 INTRODUCTION AND PURPOSE

This attachment to the SRD originally was issued as a stand-alone document (BNFL-5193-RES-01, Rev. 0, dated August 28, 1997). It has been incorporated into the SRD because it provides both background information and the basis for the radiological exposure standards reflected in the SRD Safety Criteria. In addition, it has been updated to reflect the BNFL Inc. responses to DOE Regulatory Unit questions on the Standards Approval Package.

This document is the Radiation Exposure Standard for Workers Under Accident Conditions which is a radiological safety deliverable required by the Tank Waste Remediation System Privatization (TWRS-P) Contract (DE-AC06-RL13308). This document is used by the BNFL team during the process hazards analysis (PHA) and accident analysis to ensure worker safety through identification of the need for accident prevention and mitigation features that provide worker protection against radiological and nuclear hazards. In this document, where unmodified reference is made to workers, it applies collectively to facility workers and co-located workers as defined in Sections 3.5.1 and 3.5.2 below.

The U.S. Department of Energy (DOE), in DOE/RL-96-0006, Revision 0, *Top-Level Radiological, Nuclear, and Process Safety Standards and Principles for TWRS Privatization Contractors*, (DOE-RL 1996), provides Table 1, "Dose Standards Above Normal Background." In Table 1(referred to as DOE Table 1), there are entries labeled, "To be derived," for which the contractor is to propose specific exposure standards for both facility workers and co-located workers for the following events:

- **Unlikely Events**: events that are not expected but may occur during the lifetime of the facility in the range of frequency between 10^{-2}/yr and 10^{-4}/yr (between once in 100 years and once in 10,000 years)

- **Extremely Unlikely Events**: events that are not expected to occur during the lifetime of the facility but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. Extremely unlikely events are in the range of frequency between 10^{-4}/yr and 10^{-6}/yr (between once in 10,000 years and once in 1 million years).

This document provides the required exposure standards and the bases for their selection. In addition, this document presents the BNFL approach for complying with DOE Table 1. The individual elements of this approach, as shown in Table 2-1 of SRD Safety Criterion 2.0-1 (referred to as BNFL Table 2-1), are conservative based on the requirements of the BNFL/DOE contract and, as such, satisfy the contract. For completeness, this document also discusses, and presents in BNFL Table 2-1, public exposure standards and the assumed locations of the public, facility worker, and co-located worker for use in evaluation of accident consequences and normal radioactive material releases.

2.0 EXPOSURE STANDARDS FOR FACILITY AND CO-LOCATED WORKERS

The four "To be derived" cells in DOE Table 1 have been completed by imposing a radiological exposure standard not to exceed 25 rem/event to the TWRS-P Facility or co-located workers for either unlikely or extremely unlikely events.

The 25 rem/event exposure standard for both the facility and co-located workers for unlikely and extremely unlikely events corresponds to the once-in-a-lifetime accident or emergency exposure for radiation workers which, by recommendation of the National Committee on Radiation Protection (NCRP

1963), may be disregarded in the determination of their radiation exposure status. In addition, an exposure of 25 rem/event corresponds to a conditional probability of fatality of about 2×10^{-2}. For unlikely events (defined in BNFL Table 2-1 as having a maximum occurrence frequency of 10^{-2}/yr), this equates to a maximum increase in worker lifetime risk of premature death of only 2×10^{-4}, which is considerably less than the average accidental death risk for workers in some of the safest industries (i.e., retail and wholesale trade, manufacturing, and service [EPA 1991]).

Compliance with the 25 rem/event standard is established using qualitative methods supported, where necessary, by numerical analysis that may include the development of event trees and fault trees and/or the performance of consequence analyses. From this process, preventative and mitigative engineered and administrative controls are identified.

Use of qualitative methods is consistent with the American Institute of Chemical Engineers (AIChE) guidelines (AIChE 1992), U.S. Nuclear Regulatory Commission (NRC) guidance for the performance of integrated safety analysis for 10 *Code of Federal Regulations* (CFR) 70 special nuclear material licensees (NRC 1995a), as well as DOE-STD-3009 (DOE 1994) and DOE G 420.1-X (DOE 1995). Both DOE documents state the following:

> "Estimates of worker consequences for the purpose of a safety-significant SSC designation are not intended to require detailed analytical modeling. Considerations should be based on engineering judgement of possible effects and the potential added value of safety-significant SSC designation."

Because the primary purpose of the TWRS-P Project facility and co-located worker exposure standards is to identify structures, systems, and components (SSC) required to protect these workers, the guidance cited above is both applicable and appropriate.

BNFL's principal approach for complying with the 25 rem/event worker exposure standard is the PHA. BNFL's PHA is a systematic, team-based review of the plant and treatment processes. The PHA identifies hazards and operability problems to a level of detail commensurate with the design detail available. Further hazard evaluation takes place in parallel with design development to ensure that safety continues to be built into the design process.

Having generated the list of hazards and hazardous situations, this list is subject to a further systematic team-based review where a binning process takes place. The binning process assigns postulated events to a certain hazard category and is essentially risk-based with categories of hazard defined according to a frequency/consequence matrix.

The 25 rem/event standard for unlikely or extremely unlikely events applies to events with frequencies less than 10^{-2}/yr. For those frequencies, the PHA process assigns serious and major hazardous situations as undesirable, acceptable with controls, or acceptable. For a hazardous situation to be "acceptable", its consequences must be less than 25 rem. Where there is uncertainty as to where an event should be binned (i.e., assigning a hazard category), it is binned into a higher category to ensure that the accident analysis remains conservative.

The DOE-RU has provided a guidance document (DOE-RL 1997) to be used for review of the Radiation Exposure Standard for Workers Under Accident Conditions. This guidance document includes the worker accident risk goal and the accident risk goal of DOE/RL-96-0006.

The worker accident risk goal is stated in DOE/RL-96-0006 as, "The risk, to workers in the vicinity of the Contractor's facility, of fatality from radiological exposure that might result from an accident should not be a significant contributor to the overall occupational risk of fatality to workers."

DOE/RL-97-09 (DOE-RL 1997) describes approaches that can be taken to meet this goal. The simplest approach notes that the goal can be met when (a) a worker dose standard that does not exceed 100 rem is used for extremely unlikely events (10^{-4} to 10^{-6} probability range), and (b) a worker dose standard that does not exceed 10 rem is used for unlikely events (10^{-2} to 10^{-4} probability range). For the latter probability range, the 10-rem standard relies on the assumption that the probability of accidents is evenly distributed across the probability range.

Based on experience with similar plants, BNFL considers it unlikely that the even distribution assumption will represent the actual situation for TWRS-P. Furthermore, experience indicates that there will be relatively few accidents falling into this range, and that they will be distributed toward the low probability end of the range. Consequently, a value higher than 10 rem can be used for the worker accident standard for unlikely events.

As can be seen in BNFL Table 2-1, BNFL has selected a value of 25 rem/event as the worker accident standard for both unlikely and extremely unlikely events. Because this is over 10 rem for the 10^{-2} to 10^{-4} probability range, BNFL needs to demonstrate that the worker accident risk goal is satisfied.

For the TWRS-P Project, BNFL satisfies this goal by calculating the risk of facility operation to the workers. This is a best estimate analysis based on realistic input and modeling assumptions. In performing this analysis, all structures, systems, and components capable of preventing or mitigating the event are considered. Estimates of system and component unavailabilities and unreliabilities consider failure to start and failure to run as well as maintenance-caused unavailabilities. Accident prevention and mitigation features are added to the design as necessary to satisfy the worker accident risk goal. Note 2 of BNFL Table 2-1 explicitly commits BNFL to this risk evaluation process.

The accident risk goal is stated in DOE/RL-96-0006 as, "The risk, to an average individual in the vicinity of the Contractor's facility, of prompt fatalities that might result from an accident should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed." The DOE guidance document states that a radiation exposure standard of 100 rem/event would satisfy the accident risk goal. Because the BNFL standard is 25 rem/event, the guidance document is satisfied.

In each of the four cells addressing accident exposure standards for workers and co-located workers in the unlikely and extremely unlikely events ranges, BNFL's approach does not specify an ALARA accident limit. However, Note 3 of BNFL Table 2-1 states:

> "In addition to meeting the listed dose standards for accidents, BNFL's approach to accident mitigation is to evaluate accident consequences to ensure that the calculated exposures are far enough below standards to account for uncertainties in the analysis, and to provide for sufficient design margin and operational flexibility."

This approach provides an adequate level of safety. The following paragraphs should also be noted in support of this conclusion.

BNFL's accident analyses will show compliance with exposure standards for accidents. In addition, BNFL uses a defense-in-depth approach where multiple levels of protection ensure that worker exposures from accidents will be significantly lower than calculated. This is a proven approach, considered to be effective at minimizing exposures to workers.

BNFL's approach to accident mitigation (as described in Note 3 of BNFL Table 2-1) is to examine accident consequences to ensure that calculated exposures are far enough below standards to account for uncertainties in the analysis and to provide sufficient design margin and operational flexibility. This approach is employed for all accidents (including both public and workers at all accident frequency levels) that can challenge the exposure standards, ensuring that accident exposures would be well below standards.

3.0 DEVELOPMENT OF THE BNFL APPROACH TO COMPLIANCE WITH TABLE 1 OF DOE/RL-96-0006

The overall BNFL approach to complying with DOE Table 1 is presented in this document. This approach takes the form of BNFL Table 2-1. The "To be derived" cells have been completed as discussed. The remaining cells of BNFL Table 2-1 are either identical or conservative with respect to DOE Table 1. The following sections discuss differences between DOE Table 1 and BNFL Table 2-1.

DOE Table 1 footnotes are not shown in BNFL Table 2-1. Section 2.1 of DOE/RL-96-0006 states that the footnotes refer only to the origin of the specific standards and, as such, are not considered contractual requirements unless included elsewhere in the contract.

3.1 ESTIMATED FREQUENCY OF OCCURRENCE

The second column of DOE Table 1, "Estimated Probability of Occurrence (P) (yr^{-1})," has been titled in BNFL Table 2-1, "Estimated Frequency of Occurrence (f) (yr^{-1})" because BNFL's approach is frequency based. In addition, the estimated frequency of occurrence for normal events of DOE Table 1 is redefined in BNFL Table 2-1 as any normal event regardless of frequency (nominally taken to be a frequency > 0.1/yr). The estimated frequency of anticipated events in DOE Table 1 is redefined as events with an annual frequency of occurrence of $10^{-2} < f \;\; 10^{-1}$.

With these changes, events routinely performed (e.g., melter replacement) are considered normal events rather than accidents, irrespective of frequency of occurrence. As normal events, the radiological assessment is subject to the more restrictive "per year" exposure standards rather than "per event" exposure standards. Consequently, these changes are conservative in comparison to DOE Table 1.

3.2 NORMAL EVENTS/PUBLIC AND WORKERS EXPOSURE STANDARDS

Clarifying descriptions have been included in the Normal Events/Public cell of BNFL Table 2-1 explaining that the second 100 mrem/yr standard applies to a member of the public entering the controlled area and the 25 mrem/yr standard is the public primary exposure standard for radioactive waste. The removal of DOE Table 1 footnotes (as noted above) necessitated the addition of these clarifying notes.

For the Normal Events/Worker and Normal Events/Co-located Worker cells of BNFL Table 2-1, the DOE Table 1 standard of 1.0 rem/yr ALARA design limit is replaced by a standard of 1.0 rem/yr ALARA design objective per 10 CFR 835, Section 1002(b). The corresponding worker standards for normal events in DOE Table 1 are tied to the ALARA design objectives of 10 CFR 835.1002(b) by the footnotes to DOE Table 1.

BNFL has committed to full compliance with 10 CFR 835 in the SRD, and the other sections of 10 CFR 835.1002 provide adequate requirements to ensure routine worker exposures will be ALARA. In addition, a footnote, Note 1, is included in BNFL Table 2-1. This note states the following:

> "In addition to meeting the listed design objective of 10 CFR 835.1002(b), the inhalation of radioactive material by workers and co-located workers under normal conditions is kept ALARA through the control of airborne radioactivity as described in 10 CFR 835.1002(c)."

3.3 ANTICIPATED EVENTS/WORKER AND CO-LOCATED WORKER EXPOSURE STANDARDS

References to as low as reasonably achievable (ALARA) standards have been removed for the Anticipated Events/Worker and Co-Located Worker cells of BNFL Table 2-1. The ALARA design objective of 10 CFR 835, "Occupational Radiation Protection," is applied to normal events as shown in BNFL Table 2-1. However, with the redefinition in BNFL Table 2-1 of anticipated events as those events with an annual frequency of occurrence of $10^{-2} < f \leq 10^{-1}$, the ALARA objective no longer applies because anticipated events are not part of normal operation.

This change complies fully with Section 3.2, "Radiation Protection Objective," of DOE/RL-96-0006, which states the following:

> "Ensure that during normal operation radiation exposure within the facility and radiation exposure and environmental impact due to any release of radioactive material from the facility is kept as low as is reasonably achievable (ALARA) and within prescribed limits, and ensure mitigation of the extent of radiation exposure and environmental impact due to accidents."

This aspect of BNFL Table 2-1 also represents compliance with contractual requirements because footnote 3 of DOE Table 1 references 10 CFR 835.1002(b). This section, and 10 CFR 835.202 which it references, establishes design requirements for occupational exposures other than planned special exposures and emergency exposures. Administrative limits for planned special exposures and emergency exposures are addressed in 10 CFR 835.204 and 10 CFR 835.1302 and are complied with by the TWRS-P Project.

Finally, to provide an adequate level of safety and to ensure that cost-effective safeguards affecting anticipated events are evaluated (and incorporated as appropriate) whenever the final calculated event consequence to a worker or co-located worker is 1 rem or more, BNFL's approach specifies a 1.0-rem/event design action threshold standard. In addition, a note is included in BNFL Table 2-1 to explain the application of the standard. This note (Note 4 to BNFL Table 2-1) states:

> "When a calculated accident exposure exceeds this threshold, then appropriate actions are taken. These include carrying out a less bounding (i.e., more realistic) evaluation to show that the accident consequences will be below the threshold or evaluating additional safeguards for cost-effectiveness and/or feasibility. This threshold is not a limit; it does not require the implementation of additional preventative or mitigative features if they are not both cost-effective and feasible."

3.4 EXTREMELY UNLIKELY EVENTS/PUBLIC EXPOSURE STANDARD

A standard is included in the Extremely Unlikely Events/Public cell of BNFL Table 2-1 stating that a public exposure standard target value of 5 rem/event is applied to extremely unlikely events. This target value is based on the following:

- The philosophy is that the public should be protected by a lower exposure standard than a worker. This philosophy recognizes the fact that the worker has agreed to work on the Hanford Site and has received training for avoiding hazards and dealing with hazardous situations.

- A goal to facilitate transition to the NRC as the regulatory agency with jurisdiction over nuclear safety for DOE facilities. With the exception of a 25 rem/event guideline value of 10 CFR 100 for the establishment of the exclusion area and low population zone for commercial power reactors, the NRC has not established a public exposure standard that exceeds 5 rem/event. A public exposure standard of 5 rem/event is also included in proposed rulemaking for 10 CFR 70 (NRC 1995b), which further supports the BNFL Table 2-1 value.

- With the same 5 rem/event public exposure standard for both unlikely and extremely unlikely events, there is no need to bin accidents in one of these two event frequency categories for the purpose of establishing protection of public safety.

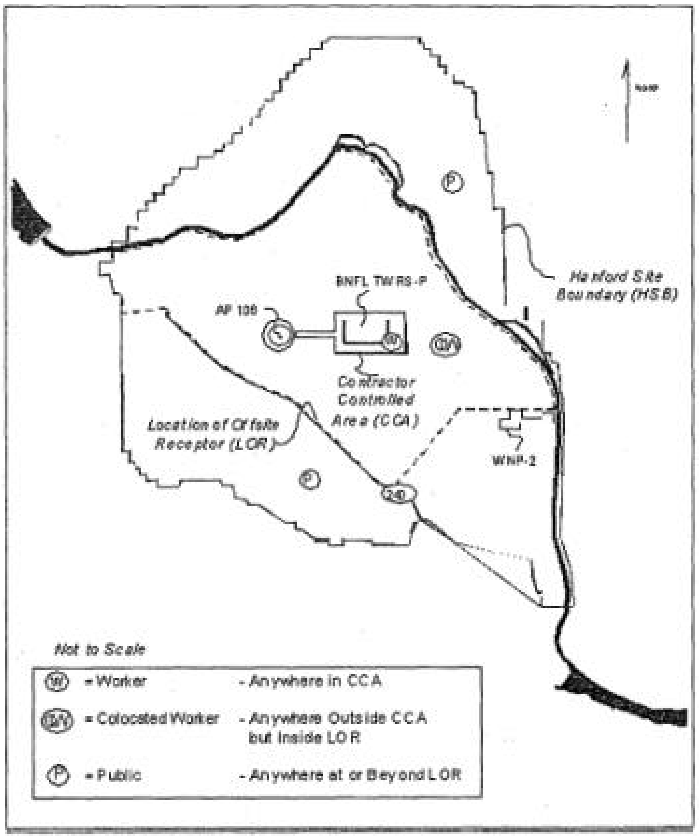
3.5 LOCATION OF RECEPTORS

In BNFL Table 2-1, a new last row has been added to clarify in DOE Table 1 of DOE/RL-96-0006 the assumed location for the facility worker, the co-located worker, and the public, for the purpose of establishing compliance with the radiological standards of DOE Table 1. The bases for the receptor locations included in this row are provided below.

3.5.1 Facility Worker

The facility worker is located at the most limiting location within the BNFL contractor-controlled area as defined in DOE/RL-96-0006, as shown in Figure D-1.

Figure D-1. Location of Facility and Co-Located Workers.



Section 6.0, "Glossary," of DOE/RL-96-0006 defines the controlled area as the following:

> "The physical area enclosing the facility by a common perimeter (security fence). Access to this area can be controlled by the Contractor. The controlled area may include identified restricted areas."

The controlled area for TWRS-P used to define the location of the facility worker, is that land leased by DOE to BNFL for the TWRS-P Project and land associated with Tank AP-106. The controlled area may include land beyond the TWRS-P Facility security fence if that fence is located within the leased area, because BNFL would have control of that area between the fence and the boundary of the leased land.

3.5.2 Co-Located Worker

Section 6.0, "Glossary," of DOE/RL-96-0006 defines the co-located worker as the following:

> "An individual within the Hanford Site, beyond the Contractor-controlled area, performing work for or in conjunction with DOE or utilizing other Hanford Site facilities."

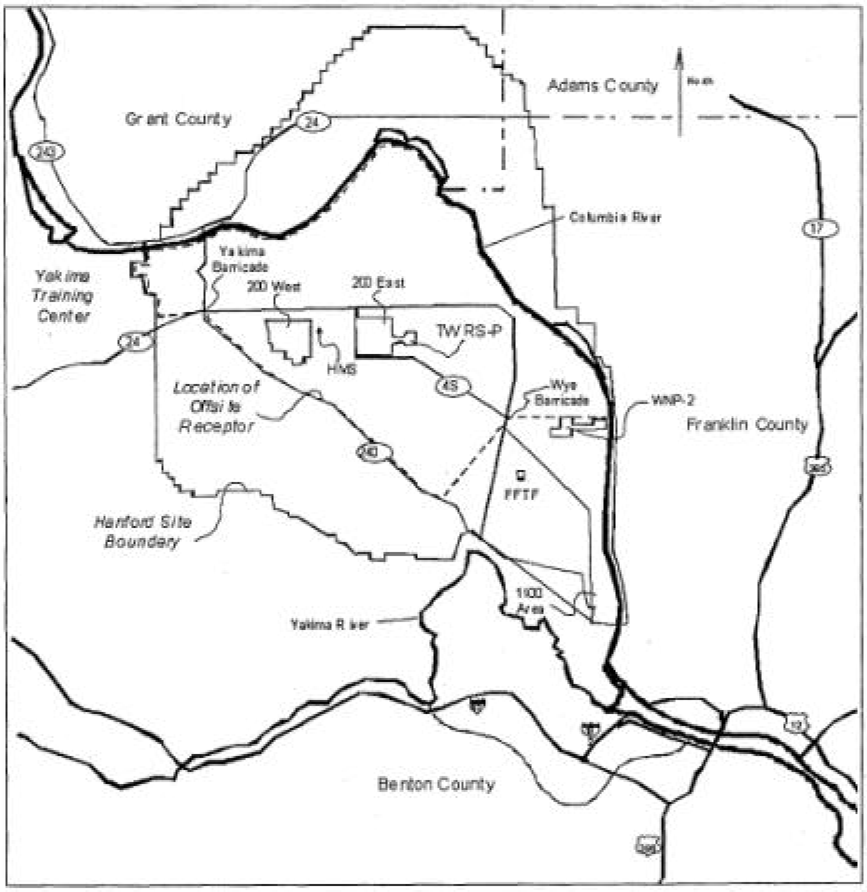
For evaluation of the TWRS-P Facility design to the exposure standards of DOE Table 1, the location of the co-located worker is either at the BNFL controlled area boundary or beyond that boundary if such a location results in higher exposure. For a ground-level release, the location of the co-located worker is considered no closer than 100 m from the release point.

3.5.3 Public

The location of the public (i.e., the offsite receptor) for the purpose of establishing compliance with the last column of DOE Table 1 of DOE/RL-96-0006, is established at the most limiting exposure location along the near bank of the Columbia River, Highway 240, and a southern boundary as shown in Figure D-12.

This area includes land for which it is reasonable to assume DOE will retain the right to control activities and limit access under accident conditions for the operating life of the TWRS-P Facility. Specifying the near river bank excludes the Columbia River for which DOE does not control activities (DOE-RL 1995). Specifying Highway 240 excludes the Arid Lands Ecology Reserve of which DOE might relinquish control during the operating life of the TWRS-P Facility. The southern boundary serves to exclude the Washington Public Power Supply System's WNP-2 commercial nuclear power plant (whose workers should be considered members of the public), and the Hanford Site 300, 400, and 1100 Areas. The 400 Area includes the Fast-Flux Test Facility.

Figure D-~~1~~2. Boundary for Location of Offsite Receptor for the Purpose of Implementing DOE/RL-96-0006, Rev. 0, Table 1, Public Exposure Standard.



In footnotes 10 and 12, DOE Table 1 of DOE/RL-96-0006 makes reference to 10 CFR 72, "Licensing Requirements for the Independent Spent Fuel (ISFSI) and High Level Radioactive Waste," and 10 CFR 100, "Reactor Site Criteria," to relate to the public exposure standards for unlikely and extremely unlikely events. While the siting requirements and guidance of Parts 72 and 100 are not applicable to the TWRS-P Facility, the requirements for establishing the location of the offsite receptor in these two cited regulations are useful for locating the offsite receptor for a waste processing facility such as TWRS-P. Section 72.106, "Controlled Area Boundary of an ISFSI or Monitored Retrievable Storage (MRS)," includes the following statements relative to the boundary to be assumed for the evaluation of radiological exposure to the public:

> "The minimum distance from the spent fuel or high-level radioactive waste handling and storage facilities to the nearest boundary of the controlled area shall be at least 100 meters."

> "The controlled area may be traversed by a highway, railroad or waterway, so long as appropriate and effective arrangements are made to control traffic and to protect public health and safety."

Title 10 CFR 100 establishes a guideline value of 25 rem for 2 hr at the exclusion area boundary. For the exclusion area, 10 CFR 100.3, "Definitions," states the following:

> "(a) *Exclusion area* means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to the facility as to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety. Residence within the exclusion area shall normally be prohibited. In any event, residents shall be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazards to the public health and safety will result."

As can be seen from the above excerpts, the assumed location for the offsite receptor for TWRS-P is consistent with 10 CFR 72 and 10 CFR 100. In addition, the proposed southern boundary takes advantage of the road junction at the Wye barricade (Figure F-1) for control of access to the site during accident conditions.

4.0 REFERENCES

10 CFR 70, "Domestic Licensing of Special Nuclear Material," *Code of Federal Regulations*, as amended.

10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," *Code of Federal Regulations*, as amended.

10 CFR 100, "Reactor Site Criteria," *Code of Federal Regulations*, as amended.

10 CFR 835, "Subpart C - Standards for Internal and External Exposure," *Code of Federal Regulations*, as amended.

AIChE, 1992, *Guidelines for Hazards Evaluation Procedures, Second Edition with Worked Examples*, Center for Chemical Process Safety, American Institute of Chemical Engineers, New York, New York.

DOE 1994, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*, DOE-STD-3009-94, U.S. Department of Energy, Washington, D.C.

DOE 1995, *Implementation Guide for Nonreactor Nuclear Safety Design Criteria and Explosives Safety Criteria*, DOE G 420.1-X, Revision G, U.S. Department of Energy, Washington, D.C.

DOE-RL 1995, *Clarification of Hanford Site Boundaries for Current and Future Use in Safety Analysis*, letter Walter B. Scott, DOE-RL to Contractors, dated September 26, 1995, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

DOE-RL 1996, *Top-Level Radiological, Nuclear, and Process Safety Standards and Principles for TWRS Privatization Contractors*, DOE/RL-96-0006, Revision 0, U.S. Department of Energy, Richland Operations Office, Richland Washington.

DOE-RL 1997, *Guidance for Review of TWRS Privatization Contractor Radiation Exposure Standards for Workers*, DOE/RL-97-09, U.S. Department of Energy, Richland Operations Office, Richland Washington.

EPA 1991, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*, U.S. Environmental Protection Agency, Washington, D.C.

NCRP 1963, *Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure*, Handbook 69, Addendum 1, National Bureau of Standards, Washington, D.C.

NRC 1995a, *Integrated Safety Analysis Guidance Document*, NUREG-1513, Draft, U.S. Nuclear Regulatory Commission, Washington, D.C.

NRC 1995b, *Preliminary Working Draft of Revision of 10 CFR 70 Updated*, 4/05/95, provided at the NRC public meeting of May 2, 1995, U.S. Nuclear Regulatory Commission, Washington, D.C.